NON-PUBLIC?: N
ACCESSION #: 9108120383
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Oconee Nuclear Station, Unit 3 PAGE: 1 OF 11

DOCKET NUMBER: 05000287

TITLE: Equipment Failure Closes Pneumatic Valve In Condensate Demineralizer System Causing Loss of Feedwater and Reactor Trip
EVENT DATE: 07/03/91 LER #: 91-007-00 REPORT DATE: 08/02/91

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)
OTHER - 50.72(b)(2)(ii)

LICENSEE CONTACT FOR THIS LER:
NAME: Henry R. Lowery, Chairman, TELEPHONE: (803) 885-3034
Oconee Safety Review Group

COMPONENT FAILURE DESCRIPTION:
CAUSE: F SYSTEM: SF COMPONENT: 65 MANUFACTURER: F180
F BA V V030
F SD XIS M225
REPORTABLE NPRDS: YES
YES
YES

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On July 3, 1991, at 1118 hours, while operating at 100 % Full Power, Oconee Unit 3 tripped as a result of a loss of feedwater. The loss was initiated when particles from a degraded seal clogged an Instrument Air flow path in the master valve controller for the condensate demineralizer system. This caused five parallel valves to fail closed, blocking all condensate flow. Demineralizer bypass valves could not open to compensate because an Operator had failed to return them to Automatic control. The loss of condensate flow resulted in the trip of condensate booster pumps due to low suction pressure, which then caused a trip of

the main feedwater pumps, followed by the reactor trip. Emergency feedwater pumps started, but a solenoid valve failed, requiring manual operator action to control flow to one steam generator. The operators stabilized the unit at hot shutdown. Corrective actions included replacement of the defective parts and counselling of the Operator. The root cause was Equipment Failure with a contributing cause of Inappropriate Action.

END OF ABSTRACT

BACKGROUND

The Condensate System EIIS:SD! has three main system functions: 1) to deliver condensate from the condenser hotwell to the suction of the main feedwater EIIS-SJ! pumps at adequate pressure, 2) to purify the condensate for chemistry control, and 3) to increase the temperature of the condensate to improve thermal efficiency. Chemistry control is largely provided by the Demineralizer System (Powdex) EIIS:SF!, which consists of five parallel resin bed filter cells EIIS:DM!. Each cell can be individually valved out of service for resin replenishment (precoating). Precoating is performed as needed on a staggered schedule to avoid having several cells out of service simultaneously. A master controller EIIS:65! provides a pneumatic control signal to valves on each cell to balance flow equally across the cells in service. Individual controllers receive this signal and compensate for differences in pressure drop over the individual cells.

Additionally, two valves, C-14 and C-15, act as Powdex bypass valves. They open sequentially in response to a single controller which can be operated either by the control room operator when in manual or in response to overall Powdex system pressure drop when in automatic.

EVENT DESCRIPTION

In March, 1991, during the last Unit 3 refueling outage, Nuclear Station Modification 32522 was implemented. This modification revised controls for several valves which formerly had reverse-acting pneumatic controllers. The old controllers showed 100 % demand for zero flow. The new controllers showed 100 % demand for full flow. As part of this modification, the controllers for several valves, including the condensate demineralizer system (Powdex) Bypass Valves (C-14 and C-15), were replaced with controllers which have programmable logic for automatic control functions. The old controller for the Powdex Bypass responded to a high pressure drop across the Powdex by opening the Bypass valves even if the controller was manually set to an intermediate valve position. The new controller was programmed so that the high pressure drop signal would only open the Bypass valves if the controller was in AUTO. An operator training package containing a description of this change was distributed prior to the end of the outage. Additionally, a label was installed immediately under the controller, stating that it should be placed in AUTO whenever the Bypass Valves were closed.

On July 3, 1991, Unit 3 was operating at 100 % full power when Operations and Chemistry personnel began the routine operation of precoating a cell in the Powdex. At 0950 hours, Control Room Operator (CRO) A placed the control for Powdex bypass valves in manual and opened them to maintain adequate Condensate flow and Condensate Booster Pump (CBP) suction pressure. At approximately 0952 hours, Chemistry Technician A (CT A) placed the Powdex master controller into manual and took one of five cells out of service to precoat the cell by closing its inlet and outlet valves.

After the cell was isolated, the master controller was returned to AUTO. CRO A used the bypass valve controller to throttle C-14 and C-15 until the other four cells carried 85 % of total condensate flow and the remaining 15 % flow was routed through the bypass. The bypass controller was left in Manual during the precoat cycle.

At 0952 hours CRO A started Low Pressure Injection (LPI) EIIS:BP! pump A for a scheduled Performance Test. After data acquisition was complete, pump A was stopped, the system was re-aligned and CRO A started LPI pump B at 1047 hours. He set the flow rate as required by the test procedure, and left the control room to perform other duties after giving verbal turnover concerning the tasks in progress to CROs B and C.

After the Powdex cell was precoated, CRO B increased Powdex Bypass flow by opening C-14 and C-15 again at 1048. CT A valved the Powdex cell back into service and placed the Powdex master controller back into AUTO. At approximately 1056 hours, CT A telephoned the control room to have C-14 and C-15 returned to normal. CRO B states that lie operated the control to close the valves, but was interrupted by a second telephone call prior to returning the control to AUTO mode. The valves finished closing at 1057 hours. CT A confirmed from local indications of individual cell flows that full condensate flow was going through the Powdex cells, concluded that the bypass valves had closed and considered the procedure step satisfied. CT A had no local indication of AUTO/MANUAL mode for the Powdex Bypass valves, and the Chemistry procedure did not specifically address having the Bypass control placed in AUTO.

CRO B states that, in response to the second telephone call, lie became involved in supporting the LPI pump test which was already in progress. His involvement continued for several minutes after which he forgot to finish returning the control for C-14 and C-15 to AUTO. At 1109 hours he secured LPI pump B and subsequently aligned and started LPI pump C at 1117 hours.

At 1117:33, alarms were received in the control room to indicate high differential pressure across the Powdex, high Hotwell Pump discharge pressure, and low CBP suction pressure. Alarms also indicated low hotwell flow and an automatic start of the standby CBP. At 1117-36 the Integrated Control System EIIS:JA! reached a feedwater (FDW)/Reactor power mismatch limit. This caused reactor power to be decreased to match a decrease in actual FDW flow. The operators investigated the alarms, but, before they could take any corrective actions, the unit tripped from 95% full power at 1118:02. A review of post trip data shows that the CBP suction pressure indicated low, which caused a trip of all three CBPs, followed rapidly by trip of both Main FDW Pumps, which, in turn, caused an anticipatory trip of the reactor.

Several immediate automatic actions occurred. All three Emergency FDW EIIS:BA! Pumps started. The Control Rod Drive EIIS:AA! breakers EIIS:BRK! opened, and all control rods were inserted into the core.

shutting down the reactor. The turbine/generator tripped, station auxiliary power EIIS:EA! switched from normal to start-up (Emergency) power, and the Main steam Relief Valves (MSRVs) and Turbine Bypass Valves opened.

The operators also took manual action. They confirmed that the reactor and turbine had tripped, verified that the Emergency FDW Pumps had started, and monitored for proper operation of other automatic equipment. They started a second High Pressure Injection (HPI) EIIS: BG! pump at 1118:36 and opened HP-26, HPI Loop A Emergency Make-up Valve to increase HPI flow to maintain Pressurizer level.

At 1118:47 CBP suction pressure increased from approximately 85 psi to 163 psi (Hotwell Pump discharge pressure). Pressure between the CBP discharge and the Main FDW Pump suction increased to 750 psi. Main FDW Pump discharge pressure momentarily increased to 1145 psi then decayed to approximately 750 psig. At 1119 both "D" heater drain pumps were manually stopped, and Main FDW pump discharge pressure dropped to approximately 165 psig.

Also at 1119, the operators shut down the Turbine Driven Emergency FDW Pump, after confirming that both Motor Driven Emergency FDW pumps were operating.

At 1120 the operators closed HP-26 and stopped the second HPI pump.

As Steam generator level dropped toward the post trip setpoint, it was observed that FDW-315, Emergency FDW throttle valve to Steam Generator A, was not controlling properly in AUTO, so CRO A took manual control at approximately 1122.

One of the Turbine Bypass Valves, MS-19, had been observed to operate erratically in Automatic following a previous Unit 3 trip on 6-9-91, so the operators took Turbine Bypass Valve control into manual at 1129.

The operators observed that the low flow alarm for cooling water flow to Motor Driven Emergency FDW rump B did not reset as expected. A Non-Licensed Operator was dispatched to check the local indication and verified that flow was indicating higher than the low flow alarm setpoint.

Two Main Steam Relief Valves did not reseat until after main steam pressure was reduced to approximately 88 to 90 % of their actuation setpoints.

Specific post-trip parameters remained in acceptable limits. Reactor Coolant System (RCS) EIIS:AB! pressure ranged between 1855 and 2202 psig. Momentary operation of the second HPi pump enabled the Operators to maintain Pressurizer inventory on scale between a high of 230 inches at the time of trip and a low of 89.5 inches. RCS temperatures converged smoothly to approximately 555 F. Steam Generator pressure reached a post-trip high of 1111 psig and was controlled at approximately 1010 psig except when

pressure was reduced to approximately 940 psig to reseat two main steam relief valves. Steam generator inventory reached a minimum of 21 inches prior to the operators taking action to compensate for the failed FDW-315.

An investigation was started to determine the exact cause of the trip. The CBP emergency low suction pressure switch, PS-228 EIIS:XIS!, was found to be leaking. It was isolated and found to have a split diaphragm. It was observed that one of four housing bolts which hold the diaphragm assembly in place was missing. The technician left the pressure switch electrical circuit open, clearing the trip signal to the CBPs, and permitting subsequent restart of a CBP. The entire switch assembly, including diaphragm, was replaced prior to unit restart. This failure was initially thought to have been the cause of the trip, because an emergency low suction pressure signal, sustained for 30 seconds, will trip all three CBPs, cause the Main FDW Pumps to trip, and result in a reactor trip. However, Transient Monitor EIIS:IQ! and control room indications show that a real flow reduction occurred prior to the trip. Also, the Hotwell Pump discharge pressure increased, indicating a flow blockage in the Powdex system.

Therefore, the Powdex was checked and the master flow controller was found to have failed, which caused the outlet valves on the individual demineralizer cells to close when they should have been open. The controller was further investigated. A pneumatic AUTO/MANUAL transfer. switch was found to have the AUTO position air port clogged by particles from a worn rubber seal. The seal was replaced.

After the unit was stabilized at hot shutdown. Operations personnel desired to restart the main FDW pumps. One of the first steps to do so required starting a CBP in condensate recirculation mode, in which condensate is routed from the Hotwell through Powdex to the Upper Surge Tank (UST). Water in the UST is then used to makeup to the Hotwell as Hotwell level drops. However, no procedure specifically covered restart of a CBP under the existing conditions. Specifically, a concern was raised that starting a CBP would cause hot water from feedwater heaters to be pumped into the UST, raising the UST temperature above 130 F. The UST also serves as the primary source of water for the Emergency FDW pumps and 130 F is the maximum supply temperature assumed in Design Engineering calculations of the required Emergency Feedwater flow rates following a trip from full power. Several procedures, including the Reactor Trip Recovery Procedure, contained caution statements that the UST temperature must be less than 130 F with the Reactor critical, but it could be up to 190 F with the Reactor subcritical. Operations Staff personnel were consulted and concurred with the decision that the 190 F limit applied because the Reactor was subcritical.

At 1330, the operators restarted a CBP and established condensate recirculation. As expected, UST temperature rose to approximately 170 F. within ten minutes. Shortly after entering this line-up, the operators observed that the UST level was indicating less than the 6 foot minimum.

level required by Technical Specifications, despite the fact that make-up flow from several sources should have been causing the level to increase. Make-up flow was increased further and the Condensate Storage Tank, which receives overflow from the UST, began to overflow onto the turbine building basement floor. This indicated that the UST was actually full, so make-up flow was then isolated. A work request was initiated to investigate the level instrumentation. Subsequent review of the Reactor Trip Recovery Procedure revealed another caution, on a later page than the first caution, which stated that the UST could not exceed 130 F unless it had been greater than five hours after the reactor trip.

The operators started a main FDW pump, and stopped the motor driven Emergency FDW pumps at 1427, terminating the loss of feedwater event. The reactor went critical at 2310 hours and the turbine was placed back on-line at 0230 hours, 7-4-91.

Analysis of Post-Trip data also showed that, although the Emergency Feedwater Pumps were started by one actuation path, i.e. due to low control oil pressure on the Main FDW Pump turbines, they did not receive a timely start signal from the other actuation path. The second path is actuated by low Main FDW Pump discharge pressure. It was observed that pressure in the condensate/feedwater system stayed higher than the low discharge pressure setpoint due to continued operation of both "D" heater drain pumps. Analysis showed this phenomenon to be a potential occurrence on loss of feedwater events for all three Oconee units, therefore one path of Emergency FDW actuation logic has been technically inoperable for some time. This situation will be reported separately as LER 269/91-09.

CONCLUSIONS

The root cause of this event is Equipment Failure. The Powdex master controller, a Foxboro model 52A "Consotrol" failed due to normal wear and age degradation of a rubber seal which is a component of a part number C123MT Switch Assembly. This failure is NPRDS reportable. This seal is internal to the controller and is not specifically inspected during calibration or routine maintenance. It has apparently been in service for a long time, possibly since initial installation prior to Unit 3 start-up.

A contributing cause for this event is Inappropriate Action, Improper Action where the proper action was chosen but execution failed because a required action was omitted. Had CRO B placed the Powdex Bypass Valve control in AUTO, the unit trip could have been avoided because the bypass valves would have been able to respond to the increased pressure drop across the demineralizer system as the Powdex cell flow control valves failed shut on loss of air. The Powdex Bypass Valve control has a label which states that the control should be placed in AUTO whenever the valves are closed. This is accomplished simply by pressing a button on the control. CRO B was knowledgeable of this requirement, but was distracted

by an interruption related to another task, and did not think about the need to return and complete this task.

Although other unit trips have occurred recently due to equipment failures, none involved the Powdex system controls and none of the corrective actions could have addressed this failure. Therefore, this event is not considered recurring.

Several additional equipment failures/problems occurred during this event, but were not causal factors of the event. Subsequent investigation by Maintenance revealed several apparent causes of some of the post-trip discrepancies.

1. A solenoid valve (SV) failure disabled the automatic control of FDW-315, the Emergency Feedwater Loop A throttle valve. The solenoid valve is normally energized but is required to operate to the de-energized position upon Emergency FDW actuation to permit automatic control. This failure is NPRDS reportable. The SV was a Valcor V-70900-21-3. This model SV is used in several other applications at the plant and has failed to operate properly in the past. Because it is normally energized and operates at an elevated temperature (approximately 250 F), Maintenance Engineering suspects that the temperature causes degradation leading to the valve sticking open when de-energized. This specific SV had been installed in the cooling water system for a Motor Driven Emergency Feedwater Pump on Dec 31, 1990. However, it was cannibalized to replace the previous SV on FDW-315, which failed during a Periodic Test of FDW-315 during the last Unit 3 refueling outage. Following that outage, plans were initiated to replace this model as the service intervals expire.

2. The Motor Driven Emergency FDW pump cooling water low flow alarm is normally received when the pump start signal is received, but normally clears as soon as the cooling water valve opens to establish flow. In this case, the instrument was found out of calibration enough that the indicated flow would not reset the alarm.

3. The reseating of two Main Steam Relief Valves at slightly lower

than desired system pressure is not considered a component failure. However, additional data evaluation will be performed following future trips to better identify the exact pressure at which the valves reseat and look for adverse trends.

4. The diaphragm on PS-228 (CBP suction pressure switch) ruptured post trip. Although CBP suction pressure went from a pre-trip value of approximately 80 psig to 160 psig after the trip, these switches were rated for 320 psig. Markings on the component indicate a date of manufacture of 1973. Maintenance records indicate that the switch has not been replaced since initial installation prior to Unit 3 startup in 1974. The mode of failure was a crack on the outer edge of the metal switch diaphragm. The presence of some rust on the

inside surface of the crack indicates that it had degraded over a period of time, then failed under higher than normal pressures. The failure of PS-228 was primarily due to age but may have been influenced by the missing bolt on the unit housing. The housing holds the switch diaphragm securely, and restrains the diaphragm from flexing at its periphery. The absence of the housing bolt could allow the diaphragm to flex at its outer edge, subjecting the area at the poi
t of failure to stress fatigue. The bolt was apparently left out after a gasket was poorly installed such that it blocked the bolt hole. It was not possible to identify when this occurred, who did it, or why the person(s) involved failed to realign the gasket so that the bolt could be properly installed. It was noted that the housing is not disassembled during routine calibration of the switch and the degradation would not have been visible without disassembly. This failure is NPRDS reportable. The switch is a Meletron model 2221-32.

5. For approximately thirty minutes during trip recovery the Upper Surge Tank (UST) level instrumentation indicated less than the minimum level required by Technical Specifications. This was due to the UST being overfilled when the Condensate recirculation line-up was established. The overfilled condition allowed water to enter the level instrument reference leg, which should be dry. This reduced the differential pressure seen by the instrument, resulting in an erroneous low reading. Because the actual UST level was full, rather than as indicated, the Technical Specification was not violated. The resulting overflow of uncontaminated water to the Turbine basement floor did not impact safety. However, the

procedural limit on UST temperature was exceeded. This temperature limit is based on Design Engineering calculations of the minimum Emergency FDW flow rates at assumed FDW supply temperatures for removal of RCS Decay Heat. The Emergency FDW system did adequately remove decay heat while the UST temperature was higher than the procedure limit. The cause of the procedure limit being exceeded is Defective Procedure, Ambiguous Information and Poor Format in that limits were presented differently on different pages.

There were no personnel injuries, radiation exposures, or releases of radioactive materials associated with this event.

CORRECTIVE ACTIONS

Immediate

1. Operations personnel took appropriate actions per the Emergency Operating Procedure and Abnormal Procedure for Loss of Main Feedwater to bring the unit to stable conditions.

Subsequent

1. The solenoid valve on FDW-315 was replaced.

2. The Motor Driven Emergency Feedwater (FDW) pump B low flow alarm was calibrated.

3. The operation of the Main Steam Relief Valves (MSRV) was evaluated and determined to be acceptable. The Post-Trip Review process has been revised to better define the MSRV performance expectations and to improve Post-trip determination of actual reseat pressures.

4. PS-228 pressure switch, including the diaphragm assembly, was replaced. Equivalent switches on Units 1 and 2 were inspected.

5. The seal on the Powdex master controller was replaced.

6. The Upper Surge Tank (UST) level instruments were checked and the reference legs dried.

7. Control Operator B has been counselled concerning his Inappropriate Action in this event.

8. A problem was discovered with the system function and setpoints for actuation of Emergency Feedwater pumps in response to loss of main feedwater as detected by low discharge pressure. That problem and the corrective actions are being reported separately as LER 269/91-09.

Planned

1. The solenoid valve model used on FDW-315 will be replaced with an improved model in all safety related applications at Oconee.

2. Instrument and Electrical (I&E) Section management will communicate to I&E technicians the potential failure mode of pressure switches with missing/loose assembly bolts.

3. The equivalent seals on Unit 1 and 2 Powdex master controllers will be inspected and replaced if necessary.

4. Operating Procedures will be revised as necessary to eliminate conflicting guidance on UST temperature limits.

5. A Station Problem Report will be initiated to revise the control logic of Powdex Bypass valves to allow them to respond to high Powdex pressure drop while in Manual.

SAFETY ANALYSIS

Failure of the Condensate Demineralizer (Powdex) master controller due to a clogged control air path resulted in a loss of control signal to the controllers for five parallel demineralizer resin tank flow control valves. These valves all went closed, as designed, resulting in the isolation of the condensate flow path. The Powdex system Bypass valves were effectively removed from service due to the inappropriate action of an operator. Had the Bypass valves been in service, they should have opened to provide an alternate flow path and to avoid a system transient. The loss of condensate flow path constituted a loss of feedwater (FDW).

Loss of FDW is an anticipated transient and is described in Section 10.4 of the Final Safety Analysis Report (FSAR). Loss of FDW initiates a reactor trip and starts the Emergency FDW system to provide decay heat and reactor coolant pump heat removal. In this event, most of the equipment and systems operated as designed to mitigate the consequences of the Loss of FDW. As expected, low Condensate Booster Pump suction pressure was detected and instrumentation tripped the Condensate Booster

Pumps. The Main FDW pumps tripped as expected. Instrumentation detected the low hydraulic oil pressure in the Main FDW pump turbine control systems and initiated the Loss of FDW trips of the Main Turbine and the Reactor and provided the start signal to the Emergency FDW System. All three Emergency FDW pumps started and the unit was stabilized at hot shutdown.

The failure of the redundant train of Loss of FDW detection logic, i.e. failure of the system to reach the actuation setpoint for Main FDW pump low discharge pressure, is described in a separate Licensee Event Report, but had no safety significance in this event.

The failure of the solenoid in the controls for FDW-315 resulted in that valve being unable to respond to control signals while in automatic. This was a single failure within the design basis of the Emergency FDW System. The operators took appropriate action to take manual control of the valve and maintained proper level in the affected steam generator throughout this event. Had the operator failed to take proper action, the affected steam generator would have boiled dry and the entire RCS heat load would have been carried by the other steam generator and emergency FDW train.

The failure of the diaphragm on PS-228 had no safety consequence. If it had occurred at some other time, independent of the condensate system transient associated with this trip, the result would have been another loss of feedwater trip. This device is one of many which can cause a unit trip due to single failure.

The overflow of the Upper Surge Tank (UST) is slightly more significant, in that it demonstrates that both trains of post-accident level monitoring instrumentation can be made inoperable by over filling the tank. The level instruments are intended to be used to verify the adequacy of the UST as a source of water for the Emergency FDW system. In some scenarios, the UST

inventory is expected to be depleted over time and the Emergency FDW pump suctions must be realigned to the Hotwell. This realignment requires that condenser vacuum be broken. If a similar loss of level indication occurred due to overfilling the UST during such an event, considerable operator time and resources could be diverted to performing unnecessary actions in order to assure an adequate source of water. However, if the erroneous indication is properly diagnosed, as occurred in this event, no adverse consequence would occur.

The operation with UST temperatures in excess of procedural limits also demonstrates the possibility of exceeding design basis assumptions that potentially could lead to technical or physical inoperability of systems or components. The temperature limit of 130 F is based on a design calculation assuring decay heat removal capability in the worst case conditions of operation with a single Emergency FDW pump immediately after a trip from full power. The limit of 190 F is intended to reflect the reduced decay heat removal requirements for the scenario where a loss of FDW occurs during startup several hours after a trip. In this case, only two hours, rather than the procedurally required five hours, had elapsed following the trip. The system was able to provide adequate flow to remove the existing decay heat load. However, two Emergency Feedwater pumps were in operation, with a third available if needed, rather than the one operating pump assumed in the design calculation. If one of the two operating Emergency Feedwater pumps was assumed to fail, the potential exists that the remaining pump might not have been able to provide the additional flow needed to maintain decay heat removal due to the higher supply temperature. In the unlikely event that a failure of one of the pumps had occurred during this time, the Turbine Driven Emergency FDW pump could have been restarted. Another option would have been alignment of Emergency FDW pumps on one of the other two Oconee units to supply cooler water from that unit's UST. Other options include use of the Standby Shutdown Facility Auxiliary Service Water System or forced cooling of the Reactor Coolant System by using the High Pressure Injection System to establish flow through the Power Operated Relief Valve. All of these options are included in appropriate procedures.

There were no releases of radioactive materials, radiation exposures, or personnel injuries associated with this event. The health and safety of the public was not affected by this event.

ATTACHMENT 1 TO 9108120383 PAGE 1 OF 1

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Oconee Nuclear Station
P.O. Box 1439
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DUKE POWER

August 2, 1991

U. S. Nuclear Regulatory Commission
Document Control Desk
Wash
ngton, DC 20555

Subject: Oconee Nuclear Station
Docket Nos. 50-269, --270, -287
LER 287/91-07

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report (LER) 287/91-07 concerning a reactor trip.

This report is being submitted in accordance with 10 CFR 50.73 (a)(2)(iv). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

H. B. Barron
Station Manager

RSM/ftr

Attachment

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